Appendix A

INTEC Contaminant Waste Disposals from 1952 through 1983

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TABLES

Appendix A

INTEC Contaminant Disposals from 1952 through 1983

A-1. BACKGROUND

This appendix provides the results of supplemental calculations for activation products, fission products, and transuranic (TRU) waste, and bulk actinide contaminants sent to the Subsurface Disposal Area (SDA) during the Historical Data Task (HDT) period (i.e., 1952 through 1983) (LMITCO 1995). Major waste streams that contained these contaminants are addressed. Data presented in the following sections are intended to augment inventory reassessments presented in Section 3 of the report main body and to be used in conjunction with Appendix C (for activation products). Best-estimate isotopic profiles are developed for major waste streams that contain known radiological contaminants. Because shipping records for the period from 1952 through 1983 have limited information, some conservative assumptions were used to calculate best estimates for fission, activation, and actinide activities sent to the SDA during the HDT period.

A-2. GENERAL DISPOSAL HISTORY

Disposals of waste from 1952 through 1983 included fission products, activation products, and actinides. Further waste streams were divided into nine categories for this period. This categorization of major streams was presented in Table 2 of this report.

Shown in Tables A-1 and A-2 are explicitly reported activities for the periods 1959 through 1970 and 1971 through 1983. Liquid acid pit disposals are not included. Because of limited information, explicitly reported net activities before 1959 were not listed in Table A-1. Analysis and estimates of actinide activities sent to the acid pits from 1954 through 1971 are presented separately in Section A-3.

Activities shown in Table A-1 are based on information from both Waste Area Group (WAG) 7 inventory databases and Form 110 shipping manifests. The key waste streams for the 1959 through 1970 period included solid actinide disposals, Vycor glass disposals, filter disposals, byproducts from a one-time-only Navy experiment, and general plant waste. Because of reporting practices in this period of interest, explicit calcine bearing filter disposals and general plant waste disposals were not always clearly differentiated from general plant waste forms in shipping manifests. Also, shipping documentation for filters disposals during this period were incomplete. Consequently, general plant waste and filter waste were combined. Appendix C shows results of analysis performed to differentiate actual filter disposals from general plant waste streams. Moreover, for the 1959 through 1970 period, only net activities (no separate isotopic profile activities) were generally reported. Also during this period, high-activity disposals were generally sent to trench areas for burial and higher activity disposals were sent to pit areas for burial.

Of the reported total 8.9E+04 Ci disposed of during this period, only a fraction of the waste activity accounted for solid actinide waste streams, general plant, and filter waste. Also only trace amounts of activation products were disposed of during this period. Consequently for this period, more than 95% of the net activity was attributed to mixed fission products (MFP). Generally, the scrap actinide waste consisted of unirradiated material with only nominal amounts of alpha activity. Most of the high-activity disposals for this period consisted mainly of fission products that contained smaller amounts of actinides

and trace amounts of activation contaminants. Two examples of such waste streams were Vycor glass and byproducts from a one-time Navy sponsored experiment.

Table A-1. Net reported activities for major waste streams from 1959 through 1970.

Disposal Date	Net activity (Ci)	Solid actinide scrap activity (Ci)	Trench Vycor glass activity (Ci)	Navy special case activity (Ci)	General plant waste with calcine filters activity (Ci)
1959	2.8E+02	—	—	—	2.8E+02
1960	8.4E+01	—	—	—	8.4E+01
1961	2.1E+02	—	—	—	2.1E+02
1962	2.0E+02	4.6E+01	—	—	1.6E+02
1963	3.6E+02	1.4E+02	—	—	2.3E+02
1964	2.3E+02	5.0E-05	—	—	2.3E+02
1965	1.7E+02	—	—	—	1.7E+02
1966	1.9E+04	2.2E+02	1.9E+04	—	2.3E+02
1967	1.5E+04	1.0E-01	1.5E+04	—	3.3E+02
1968	1.1E+04	1.3E+02	1.1E+04	—	1.0E+02
1969	4.2E+04	1.6E-05	1.8E+03	4.0E+04	5.2E+02
1970	1.4E+02	2.0E-02	5.0E+01	—	8.6E+01
Total	8.9E+04	5.3E+02	4.6E+04	4.0E+04	2.6E+03

From 1971 through 1983, reporting methods transitioned to more detailed breakdowns of isotopic contaminants for SDA disposals. Net reported activities for this interval are present in Table A-2. A total of 5.2E+05 Curies were sent to the SDA during this period. Yearly activities shown in Table A-2 are based on information from both Radioactive Waste Management Information System (RWMIS) data and Form 110 shipping manifests. Low-activity shipments were typically buried in pit disposal areas and high-activity shipments were sent to either trench or soil vault disposal areas.[a] Soil vault disposal activity did not begin until 1977.

Several new kinds of waste streams were sent to the SDA during this period. Disposals of neutron-activated subassembly hardware during this time frame resulted in significant increases in reported net yearly activities mainly in 1973 and 1982 (See Appendix C for additional details). The principal contaminants in this waste stream were activation products. Also, CPP-603 basin sludge and Tank Farm contaminated soil also were significant contributors to the total fission product activities during this period.

a. In some cases, large, heavily shielded containers of high-activity CPP-603 basin sludge were sent to pit locations.

Table A-2. Net reported activities for major waste streams from 1971 through 1983.

Year	Total (Ci)	Solid actinide scrap (Ci)	CPP-603 Fuel storage basin sludge (Ci)	Subassembly hardware (Ci)	Tank Farm Contaminated dirt (Ci)	Waste[a] calciner filters (Ci)	General plant waste (Ci)	Special case radium (Ci)
1971	1.1E+02	2.5E-01	—	—	—	4.0E+01	6.9E+01	—
1972	1.6E+02	1.0E+00	—	—	—	1.4E+01	1.5E+02	—
1973	3.3E+05	2.2E-02	—	3.3E+05	—	1.1E+03	5.1E+01	—
1974	5.7E+03	2.0E-03	—	—	3.9E+03	1.5E+03	2.5E+02	—
1975	9.5E+03	1.4E+01	—	4.9E+03	1.6E+03	3.0E+03	2.0E+01	—
1976	1.1E+03	8.5E-02	—	—	5.0E+01	9.9E+02	9.8E+01	—
1977	6.4E+03	1.0E-01	3.88E+03	—	1.1E+02	1.7E+03	7.5E+02	—
1978	1.1E+04	—	5.65E+03	—	2.6E+01	3.8E+03	1.8E+03	—
1979	4.4E+03	—	9.98E+02	—	—	2.8E+03	6.3E+02	1.0E-01
1980	1.9E+04	8.9E-01	1.56E+04	—	—	2.6E+03	3.0E+02	—
1981	1.7E+04	—	—	1.40E+04	2.2E-02	2.9E+03	3.7E+02	—
1982	1.1E+05	7.0E-01	—	1.15E+05	—	6.9E+00	1.7E+02	—
1983	3.0E+02	2.0E-02	—	—	—		3.0E+02	—
Total	5.2E+05	1.7E+01	2.6E+04	4.6E+05	5.7E+03	2.0E+04	4.9E+03	1.0E-01

a. Includes other miscellaneous filters.

It should be noted that the reported values in Tables A-1, A-2, and A-3 are based on Cs-137 activities. Tables that are based on all isotopes that appear later in this appendix tend to produce higher values than the original reported values.

Yearly RWMIS isotopic profile activity data are presented in Table A-3 for the 1971 through 1983 period. In some circumstances, reported isotopic profiles of disposals were limited to more generic descriptions such as MFP, mixed activation products (MAP), and unidentified gamma-beta emitters (UN-ID-B+G). For circumstances where these generic definitions were applied to waste stream activities, other assumptions and scaling factors were applied to estimate actual contaminants of concern (COC) contents.

Table A-3. SDA contaminant inventory disposals from 1971 through 1983.

Nuclide	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	Total
CO-60	—	—	1.8E+05	1.5E+01	3.3E+03	1.4E-01	6.9E+00	2.0E+01	3.8E+00	—	1.4E+02	1.1E+03	—	1.9E+05
SR-90	1.6E+01	6.2E+00	1.6E+02	1.4E+03	1.7E+02	1.4E+02	1.9E+02	1.0E+03	3.9E+02	1.1E+03	3.0E+02	1.5E+01	2.6E+01	5.0E+03
CS-137	3.5E+01	6.2E+00	1.6E+02	1.1E+03	5.0E+02	1.8E+02	5.2E+02	2.6E+03	9.1E+02	9.7E+03	3.6E+02	1.5E+01	2.6E+01	1.6E+04
Eu-152	—	—	—	—	1.1E+00	4.4E-01	1.7E+02	1.2E+02	5.4E+00	1.0E+02	—	—	—	4.0E+02
Eu-154	—	—	—	4.2E+01	3.5E+01	1.2E-01	8.4E+01	9.8E+01	4.9E+00	1.0E+02	—	—	2.0E-03	3.7E+02
RA-226	—	—	—	—	—	—	—	—	1.0E-01	—	—	—	—	1.0E-01
TH-232	—	1.1E-07	—	—	1.1E-02	9.8E-08	3.4E-04	1.3E-04	—	1.2E-05	—	—	—	1.1E-02
U-233	—	—	—	—	—	5.7E-05	—	—	—	—	—	—	—	5.7E-05
U-234	—	—	—	—	1.8E-04	6.2E-04	2.8E-02	7.7E-03	1.0E-02	1.2E-01	—	—	—	1.6E-01
U-235	2.0E-04	6.4E-04	1.1E-03	1.4E-04	9.2E-04	3.2E-05	1.7E-03	3.7E-04	6.3E-04	4.5E-03	2.4E-05	3.1E-06	—	1.0E-02
U-236	—	—	—	—	2.3E-06	3.2E-05	2.0E-04	1.3E-04	3.1E-04	3.3E-03	—	—	—	4.0E-03
U-238	1.5E-03	4.5E-04	2.1E-02	1.9E-03	7.5E-05	1.2E-05	1.1E-04	3.2E-03	5.4E-03	7.7E-04	4.1E-04	1.1E-03	1.5E-05	3.6E-02
Pu-238	—	—	—	—	—	—	2.3E-01	3.0E-02	4.2E-02	7.7E-01	—	—	2.0E-02	1.1E+00
PU-239	3.7E-01	—	1.5E-01	—	1.9E-04	—	1.4E-01	2.9E-02	3.1E-02	3.2E-01	1.4E-03	—	1.2E-03	1.0E+00
PU-240	7.2E-02	—	—	—	1.1E-05	—	7.6E-03	2.6E-03	7.3E-04	8.8E-03	3.7E-03	—	—	9.5E-02
Pu-241	9.8E+00	—	—	—	—	—	—	—	8.2E-02	1.3E+00	—	—	—	1.1E+01
Pu-242	—	—	—	—	—	—	—	—	4.6E-06	8.0E-05	—	—	—	8.4E-05
Np-237	—	—	—	—	—	—	—	—	—	—	—	—	2.1E-05	2.1E-05
Am-241	—	—	—	—	—	—	—	—	2.1E-04	—	—	—	1.6E-02	1.6E-02
Cf-252	—	—	—	—	—	—	—	—	—	—	—	—	2.1E-03	2.1E-03
MAP	—	—	2.3E+04	8.0E-02	9.2E-01	—	—	—	6.6E-02	—	—	—	—	2.3E+04
MFP	1.7E+01	1.3E+01	1.0E+02	3.0E+02	2.7E+02	—	—	2.9E+02	2.4E+02	1.0E+02	2.9E+01	2.7E+01	1.1E+01	1.4E+03
UN-ID-B+G	2.7E+00	3.5E-01	—	—	—	—	—	—	—	—	—	—	—	3.1E+00

Table A-4 is a comparison of explicit shipping records' contaminant activities combined from Tables A-1 and A-3 with contaminants reported from the comprehensive inventory report (LMITCO 1995).[b] The purpose of this comparison is to show limitations amongst shipping record disposal activities along with previous isotopic profile analysis. The principal limitations in these profiles is that a number of COCs have either not been reported or are based on outdated assumptions. Upgrades to previously reported isotopic profiles are developed in the remainder of this appendix.

Table A-4. Comparisons of known shipping records activity with previous isotopic profile analysis of contaminants sent to the SDA.[a]

Nuclide	Known Shipping Records (Ci)	HDT Best Estimate Inventory (Ci)
C-14	—	4.3E+01
Ni-59	—	1.6E+02
Ni-63	—	2.5E+04
Co-60	1.9E+05	2.0E+05
Sr-90	5.0E+03	2.0E+04
Tc-99	—	3.0E-02
Cs-137	1.6E+04	4.2E+04
Eu-152	4.0E+02	2.4E+02
Eu-154	3.7E+02	2.9E+02
Ra-226	1.0E-01	—
Th-232	1.1E-02	—
U-233	5.7E-05	—
U-234	1.6E-01	4.8E+00
U-235	1.0E-02	1.6E-01
U-236	4.0E-03	4.0E-03
U-238	3.6E-02	6.6E-01
Pu-238	1.1E+00	1.0E+00
PU-239	1.0E+00	4.8E-01
PU-240	9.5E-02	1.0E-02
Pu-241	1.1E+01	—
Pu-242	8.4E-05	1.0E-01
Np-237	2.1E-05	—
Am-241	1.6E-02	—
Cf-252	2.1E-03	—
MAP	2.3E+04	—
MFP	9.0E+04	—
UN-ID-B+G	3.1E+00	—
a. LMITCO 1995	—	—

b. Alpha activities were generally not explicitly reported before 1970 in shipping manifests. Conversions of reported weights of actinides to equivalent activities are given in the following sections of this appendix to correctly update actinide activities.

This previous analysis employed isotopic scaling factors that estimated the isotopic contents of generic amounts of reported amounts of MAP, MFP, and UN-ID-B+G.[c] However, that older analysis did not account for all required COCs because of different reporting requirements at that time. In many cases, the previous analysis focused on estimating short half-lived contaminants (important for short-term handling and transportation safety issues) rather than long-lived contaminants needed for long-term risk analysis issues. In addition, more updated isotopic scaling factors have become available since that inventory report was published. These updated scaling factors will be applied in the reassessment of isotopic profiles presented in the remainder of this appendix.

A-3. ACID PIT DISPOSAL INVENTORIES

This section augments the reassessment analysis of Acid Pit disposals documented in Section 3.4 of this report (Jorgensen et al. 1994). The Acid Pit was originally excavated to the top of the basalt bedrock and covers approximately 20,490 ft^2 (197 × 104 ft). The excavation was covered with 1 to 2 ft of soil. Lime was added periodically to neutralize the acids disposed of in the pit. During operations, a soil cover was applied over the waste at daily or weekly intervals. After the waste was emplaced, the pit was backfilled with approximately 3 ft of soil and an overlaying vegetation layer was added.

A portion of the liquid wastes sent to SDA consisted of raffinates from fuel reprocessing cold line tests. Raffinates generated in these cold line tests used unirradiated fuel that produced byproducts containing aqueous acids solutions and organic solvents. These tests were conducted before beginning a new fuel processing campaign to ensure that the uranium recovery efficiency was optimized. These solvents may have included trace amounts of fission products since cold-line tests were performed with processing hardware that contained some residual contaminants from other reprocessing campaigns of irradiated fuel. Another portion of the liquid wastes consisted of diluted sludge sent from CPP-603. Some of this sludge material was reported as being diatomaceous earth that possibly originated from filtration systems in the CPP-603 fuel storage basins.

The following assumptions were made to estimate baseline isotopic distributions for known Acid Pit disposals. U-234 was approximated to as 1% of the U-235 present for enriched uranium. U-235 was assumed to constitute 93% of the enriched uranium (Walker, Parrington, and Feiner 1989), 0.7% of the natural uranium (Walker, Parrington, and Feiner 1989), and 0.007% of the depleted uranium (99% depletion). U-238 was assumed to represent 7% of enriched uranium (93% enrichment), 99.3% of natural uranium, and essentially all of depleted uranium. Specific activities were then applied, which are 6.3E-03, 2.2E-06, and 3.4E-07 Ci/g for U-234, U-235, and U-238, respectively. In some cases, the determination of uranium isotopic compositions was based on professional judgments since not all shipment explicitly reported whether disposals consisted of depleted or natural uranium.

By applying the above assumptions to known disposal manifest information, yearly estimates of lower-bound, best-estimate, and upper-bound activities were calculated (see Tables A-5 through A-7). The activities in Table A-1 are baseline or lower-bound estimates derived from the shipping available shipping records. The known associated acid pit disposals are listed in Appendix D. Finally, because of reporting uncertainties, upper and lower-bound activities were calculated by applying an uncertainty factor of 1.5 to the baseline activities in Table A-1.

Table A-5. Lower-bound activities for disposals of uranium isotopes by year from INTEC into the Acid Pit at the RWMC.

c. The reported 8.9E04 Ci of net activity for the 1959–1970 period was approximated as being equivalent to 8.9E+04 Ci of MFP since actinide and activation products were small.

Year	U-234 (Ci)	U-235 (Ci)	U-238 (Ci)
1954	1.27E-02	4.47E-04	1.80E-02
1955	2.27E-02	8.00E-04	4.20E-02
1956	6.33E-03	2.20E-04	4.60E-03
1957	1.13E-03	4.00E-05	8.67E-04
1958	1.00E-03	3.40E-05	4.73E-04
1959	1.07E-04	3.73E-06	8.00E-05
1960	0.00E+00	0.00E+00	0.00E+00
1961	0.00E+00	0.00E+00	0.00E+00
1962	2.33E-04	8.00E-06	1.73E-04
1963	0.00E+00	0.00E+00	0.00E+00
1964	1.67E-05	5.87E-07	7.33E-09
1965	0.00E+00	0.00E+00	0.00E+00
1966	1.53E-04	5.33E-06	6.40E-08
Total	4.40E-02	1.53E-03	6.60E-02

Table A-6. Best estimate for disposals (Ci) of uranium isotopes by year from INTEC into the Acid Pit at the RWMC.

Year	U-234	U-235	U-238
1954	1.90E-02	6.70E-04	2.70E-02
1955	3.40E-02	1.20E-03	6.30E-02
1956	9.50E-03	3.30E-04	6.90E-03
1957	1.70E-03	6.00E-05	1.30E-03
1958	1.50E-03	5.10E-05	7.10E-04
1959	1.60E-04	5.60E-06	1.20E-04
1960	0.00E+00	0.00E+00	0.00E+00
1961	0.00E+00	0.00E+00	0.00E+00
1962	3.50E-04	1.20E-05	2.60E-04
1963	0.00E+00	0.00E+00	0.00E+00
1964	2.50E-05	8.80E-07	1.10E-08
1965	0.00E+00	0.00E+00	0.00E+00
1966	2.30E-04	8.00E-06	9.60E-08
Total	6.60E-02	2.30E-03	9.90E-02

Table A-7. Upper-bound estimate for disposals (Ci) of uranium isotopes by year from INTEC into the Acid Pit at the RWMC.

Year	U-234	U-235	U-238
1954	2.90E-02	1.00E-03	4.00E-02
1955	5.10E-02	1.80E-03	9.40E-02
1956	1.40E-02	5.00E-04	1.00E-02
1957	2.60E-03	9.00E-05	1.90E-03
1958	2.20E-03	7.70E-05	1.10E-03
1959	2.40E-04	8.40E-06	1.80E-04
1960	0.00E+00	0.00E+00	0.00E+00
1961	0.00E+00	0.00E+00	0.00E+00
1962	5.20E-04	1.80E-05	3.90E-04
1963	0.00E+00	0.00E+00	0.00E+00
1964	3.80E-05	1.30E-06	1.60E-08
1965	0.00E+00	0.00E+00	0.00E+00
1966	3.50E-04	1.20E-05	1.40E-07
Total	1.00E-01	3.50E-03	1.50E-01

A-4. ACTINIDE SHIPPING DISPOSAL INVENTORIES

This section augments the reassessment analysis of solid-actinide-bearing waste streams documented in Section 3.5 of this report. Trace amounts of possible actinide contaminants that may have been present in some general plant waste disposals are not directly factored into the analysis in this section. Also, actinide contaminants in other major waste streams identified in Table 2 of the main body of this report are separately discussed in this appendix.

Shipping manifests demonstrate that disposals of both of unirradiated and some irradiated actinide solid waste were explicitly identified in shipping records for the period from 1962 through 1983. These materials included dissolved fuel samples that were stabilized in solid media like vermiculite, solid fuel scrap, sporadic miscellaneous irradiated fissile material, and larger bulk disposals of uranium and thorium bearing materials. Key bulk-waste streams included fuel mockup disposals that consisted of thorium, depleted or natural uranium the CPP-601 and enriched fuel scrap from laboratory support facilities such as CPP-627. Some of this fuel scrap material was from other facilities. Examples of such scrap disposals included EBR-I fuel from ANL-W and fuel pellets from PBF.

A baseline of available reported year-by-year breakout of net actinide disposals is presented in Table A-8. The yearly disposals were calculated using spreadsheet numbers from both the WAG 7 inventory databases. In some cases, crosschecking Form 110s disposal manifest against these databases showed that some corrections were required. In particular, not all transshipped disposals were recorded in these databases.

Table A-8 also contains some plutonium disposals along with trace amounts of other transuranic contaminants for some shipments containing irradiated fissile material. A total 790 kg of actinide heavy metal was explicitly reported in these actinide waste streams. Additionally, about 500 Ci of net activity

was reported in conjunction with this waste stream (Table A-8). However this activity was mainly attributed to fission contaminants that occurred in a very limited number of shipments sent to the SDA in the 1960s. Because of the much larger volumes of fission activity reported in other waste streams (See remaining sections of Appendix A), these fission contaminants have been neglected in the analysis presented in this section of Appendix A.

Categorization of disposals based on actinide enrichment types is presented in Table A-9. Net disposals of heavy metal contaminants for this waste stream included reported amounts consisting of 4.6E02 kg of natural uranium, 2.3E+02 kg of depleted uranium, 4.4 kg of enriched uranium, and 1.0E+02 of thorium. In some circumstances, shipments identified the specific waste types; in other circumstances, limited information required that enrichment levels had to be inferred.

Table A-8. Baseline for reported yearly actinide weights sent to the SDA from 1962 through 1983.

Disposal Year	U-235 (gm)	U-238 (gm)	Pu-239 (gm)	Th-232 (gm)	Net HM Weight (gm)	Total Activity (Ci)
1962	1.1E+02	1.9E+03	—	—	2.1E+03	4.6E+01
1963	1.1E+03	—	—	—	1.1E+03	1.2E+02
1964	4.4E-01	—	—	—	4.4E-01	5.0E-05
1965	—	—	—	—	—	—
1966	5.1E+01	1.7E+02	—	—	2.2E+02	2.2E+02
1967	1.3E+03	1.9E+05	—	—	1.9E+05	1.0E-01
1968	9.3E+02	1.2E+05	—	—	1.2E+05	1.3E+02
1969	4.1E+00	4.5E+00	—	—	8.6E+00	1.6E-05
1970	1.9E+03	3.0E+05	—	—	3.0E+05	2.0E-02
1971	1.9E+02	4.2E+03	3.0E+00	—	4.4E+03	2.5E-01
1972	1.4E+03	3.0E+02	—	1.0E+00	1.7E+03	1.0E+00
1973	6.4E+04	3.0E+02	—	—	6.4E+04	2.2E-02
1974	6.7E+01	5.7E+03	—	—	5.8E+03	2.0E-03
1975	4.3E+02	2.2E+02	—	1.0E+05	1.0E+05	1.4E+01
1976	6.9E+00	3.2E+01	—	—	3.9E+01	8.5E-02
1977	4.6E+02	3.1E+01	—	3.1E+03	3.6E+03	1.0E-01
1978	—	—	—	—	—	—
1979	—	—	—	—	—	—
1980	—	—	—	1.0E+02	1.0E+02	8.9E-01
1981	—	—	—	—	—	—
1982	1.4E+00	3.2E+03	—	—	3.2E+03	7.0E-01
1983	—	4.5E+01	2.0E-02	—	—	—
Total (gm)	7.2E+04	6.2E+05	3.0E+00	1.0E+05	7.9E+05	5.4E+02

Table A-9. Baseline shipping inventory for categories of actinide waste sent to the SDA from 1962 through 1983.

Year	NU (gm)	DU (gm)	EU (gm)	Thorium (gm)
1962	2.0E+03	—	9.9E+01	—
1963	—	—	1.1E+03	—
1964	—	—	4.4E-01	—
1965	—	—	—	—
1966	—	1.7E+02	5.1E+01	—
1967	1.9E+05	—	—	—
1968	1.2E+05	—	1.5E+02	—
1969	—	—	8.6E+00	—
1970	1.5E+05	1.5E+05	1.4E+02	—
1971	—	4.1E+03	2.3E+02	—
1972	—	—	1.7E+03	—
1973	—	6.4E+04	3.6E+01	—
1974	—	5.7E+03	6.7E+01	—
1975	—	1.2E+02	4.4E+02	1.0E+05
1976	—	—	3.9E+01	—
1977	—	—	3.6E+02	3.2E+03
1978	—	—	—	—
1979	—	—	—	—
1980	—	—	—	1.0E+02
1981	—	—	—	—
1982	—	3.2E+03	—	—
1983	—	4.5E+01	—	—
Total (kg)	4.6E+02	2.3E+02	4.4E+00	1.0E+02

The baseline reported actinide waste streams in Tables A-8 and A-9 were used to generate best estimates of net actinide contaminant activities sent to the SDA during the HDT. Isotopic distributions of shipments given in Table A-8 were developed using Tables A-10 and A-11. The distributions of U-238, U-235, and U-234 for different enrichments are dependent on processing histories. For depleted uranium the distribution of U-238 and U-235 are 99.8% and 0.2%, respectively. For natural uranium, the distribution of U-238 and U-235 are 99.3 and 0.7%, respectively. For natural uranium and depleted uranium, the ratio of U-234 to U-235 was set at 0.8% (Benedict et al. 1981). For enriched disposals, the fraction of U-234 was conservatively estimated to be 1% of the U-235 mass (Rucker and Johnson 1998). Trace amounts of U-236 along with other unreported amounts of TRU contaminants may have been

present in some shipments.[d] However, these contaminants were neglected because they were judged to be a small fraction of the net amounts present in other major waste streams.

To calculate best-estimate contaminant activities, assumptions about uncertainties associated with the above solid actinide waste streams were made. From Section 3.5 of this report a generic uncertainty factor of 1.5 was employed to estimate best-estimate and upper-bound activities associated with solid waste streams. By using the conversion factors and distributions from Tables A-10 and A-11 the net activities for the HDT period were calculated in Table A-12. The lower-bound activities in Table A-12 were defaulted to the known reported baseline actinide masses using reported weights from Tables A-8 and A-9 and using conversion factors from A-10 and A-11. The best-estimate and upper-bound activities were calculated from the lower-bound activities with the 1.5 and 2.0 uncertainty multipliers. The general rationale for using these factors is discussed in Section 3.14 of this report. Other waste stream dependent reasons for using these uncertainty factors revolve around current limitations in disposal documentation that include:

- Possible incomplete documentation of actinide disposals shipped from such outside facilities like Atomics International.
- Incomplete documentation of solid waste disposals that may have contained significant amounts of accounted-for actinides in the 1950s.
- Possible unreported actinide disposals that were reported as general plant waste or collocated with low-activity general plant waste.

Table A-10. Nuclide conversions factors for estimating corresponding activities of solid based actinide wastes.

Nuclide	Conversion (Ci/g)
Th-232	1.1E-07
U-234	6.3E-03
U-235	2.2E-06
U-238	3.4E-07
Pu-239	6.2E-02

d. Later shipments containing unirradiated EU may have contained U-236, if this EU consisted of reprocessed fuel.

Table A-11. Nuclide weight distributions for different uranium enrichments.

Depleted Uranium Isotopic Mass Distribution	Mass Fraction
U-235	0.2%
U-238	99.8%
U-234/U-235 mass fraction	0.8%
Natural uranium isotopic mass distribution	—
U-235	0.7%
U-238	99.3%
U234/U235 mass fraction	0.8%
Enriched fuel isotopic mass distribution	—
Total U235	Can be as high as 93% of total weight
Total U238	Varies with U235 fraction
U234/U235 mass fraction	1.0%

Table A-12. Total solid actinide activity disposals for the HDT period.

Nuclide	Lower-bound Activity (Ci)	Best-estimate Activity (Ci)	Upper-bound Activity (Ci)
U-234	2.9E-01	4.3E-01	6.4E-01
U-235	1.1E-02	1.7E-02	2.5E-02
U-238	1.5E-01	2.3E-01	3.4E-01
Pu-239	1.2E-01	1.9E-01	2.8E-01
Th-232	7.3E-03	1.1E-02	1.6E-02

A-5. ESTIMATED CONTAMINANTS FOR VYCOR GLASS DISPOSALS

This section augments the reassessment analysis of leached Vycor glass bearing waste streams documented in Section 3.6 of this report. This stream consisted of significant amounts of fission contaminants along with actinide contaminant residues embedded in the Vycor glass material. The Vycor glass was used as molds for the casting new EBR-II fuel pins. The casting operation was conducted at ANL-W. Shipments of Vycor scrap were made from ANL-W to INTEC for reprocessing from July 1965 to December 1969. Additionally, it was estimated that approximately 610 kg of Vycor scrap containing 80 kg of fuel alloy material went to INTEC for reprocessing (Stevenson 1987).

At INTEC, the Vycor glass was then leached with hot nitric acid to recover embedded uranium residues (estimated to be about 80 kg of enriched uranium). It was estimated that 80 to 90% of the embedded uranium was recovered in the leaching process (Stevenson 1987). The remaining leached glass waste still contained substantial amounts of fission products with some remaining actinides and TRU residues before burial in the SDA. The net amount of fuel scrap in these shipments was estimated to be in

the range of 8-16 kg after the leaching process was completed. About 4 tons of new fuel pins were manufactured with the Vycor glass casting process.

The Vycor glass was shipped in 10-in.-high metal cans to INTEC for processing. These cans were put in lead-shielded 3-ton casks. After the uranium leaching process was finished the treated glass was returned to the same metal can and the can was resealed and buried. During the latter period of the reprocessing campaign, these metal cans were supplied with stainless steel liners at ANL-W before shipping to INTEC. Vycor glass disposals to the SDA began in 1966 and ended in 1970. Shown in Table A-13 is a yearly summary of reported activities. The net total activity for this waste stream amounted to 4.6E4 Ci. The total was based on known shipments of Vycor glass that were reported on Form 110 shipping manifests.

Table A-13. Summary of reported yearly activities for Vycor glass disposals.

Year	Activity (Ci)
1966	1.9E+04
1967	1.5E+04
1968	1.1E+04
1969	1.8E+03
1970	5.0E+01
Total	4.6E+04

The actual method for estimating reported disposals activities for the shipments in Table A-13 was unknown. For the purpose of the reassessment analysis, it was assumed that Cs-137 was the dominant is gamma-emitting isotope measured in these waste shipments. It was assumed that sufficient time had passed to allow other shorter half-lived gamma emitting isotopes to become insignificant. This assumption was judged as reasonable since expended fuel from EBR-II was not immediately reprocessed.

In order to reassess the contaminant profiles in these streams, it was assumed that the relative amounts of fission products were disproportionately larger than the corresponding actinide contaminants. It was judged that the higher mobility fission products were more likely to adhere to the Vycor glass molds as opposed to solidified metallic actinide fuel. To develop updated contaminate distributions for this waste stream, calculations for spent EBR-II Mark-I fuel were employed (Wenzel 2000a).

Shown in Table A-14 are net activities per unit fuel element, the associated scaling factors. In the second column of Table A-14, the fission products were scaled to Cs-137 and the actinides and actinide decay products were scaled to U-235.[e] The ORIGEN2 based calculations in Table A-14 were based on early Mark I EBR-II fuel designs with approximately 2% burnup of U-235 and 52% enrichment (Croft 1980). However, during the ANL-W fuel reprocessing campaign, some fuel burnups were as high as 6% and enrichments as high as 67%.[f] Shown in the third column are the calculated baseline activities. The fission products were calculated by assuming that Cs-137 was the dominant gamma emitter so that the all

e. As such, all of the fission products were scaled to Cs-137 (all isotopes from H-3 to Eu-154) and all actinides and decay daughters were scaled to U-235 (all isotopes from Pb-210 to Cm-246).

f. The higher burnup fuels were incorporated into Mark-II fuel designs.

fission products were scaled to 4.6E4 Ci. The proportionately lower actinide activities were scaled to 8 kg of fuel at 52% U-235 enrichment. The highlighted portions of Table A-14 correspond to the nuclides scaled to U-235 activities.

Table A-14. Mark-IA EBR-II isotopic and scaling factor distributions.

Nuclide	Activity per Fuel Element (Ci)	Scale Factor Normalized to Cs-137 and U-235	Baseline Activities (Ci)
H-3	1.1E+00	7.3E-03	3.4E+02
Be-10	4.6E-09	3.1E-11	1.4E-06
C-14	1.8E-07	1.2E-09	5.7E-05
Sr-90	1.4E+02	9.2E-01	4.2E+04
Nb-94	2.7E-08	1.8E-10	8.4E-06
Tc-99	2.1E-02	1.4E-04	6.7E+00
I-129	5.8E-05	3.9E-07	1.8E-02
Cs-137	1.5E+02	1.0E+00	4.6E+04
Eu-152	7.9E-04	5.3E-06	2.5E-01
Eu-154	1.3E-01	8.7E-04	4.0E+01
Pb-210	2.2E-10	3.6E-08	3.2E-10
Ra-226	5.6E-09	8.9E-07	8.0E-09
Ra-228	3.6E-14	5.8E-12	5.2E-14
Ac-227	3.2E-08	5.1E-06	4.6E-08
Th-228	2.5E-05	4.0E-03	3.6E-05
Th-229	6.8E-10	1.1E-07	9.9E-10
Th-230	6.5E-06	1.0E-03	9.4E-06
Th-232	2.2E-13	3.5E-11	3.1E-13
PA231	5.2E-07	8.4E-05	7.5E-07
U-232	3.5E-05	5.6E-03	5.0E-05
U-233	1.3E-06	2.1E-04	1.9E-06
U-234	1.8E-01	3.0E+01	2.7E-01
U-235	6.2E-03	1.0E+00	9.0E-03
U-236	1.2E-03	2.0E-01	1.8E-03
U-238	8.8E-04	1.4E-01	1.3E-03
Np-237	9.2E-05	1.5E-02	1.3E-04
Pu-238	1.2E-02	1.9E+00	1.7E-02
Pu-239	4.9E-01	7.9E+01	7.1E-01
Pu-240	3.7E-03	5.9E-01	5.3E-03
Pu-241	2.2E-03	3.5E-01	3.2E-03
Pu-242	9.2E-11	1.5E-08	1.3E-10
Pu-244	1.1E-21	1.7E-19	1.6E-21
Am-241	1.2E-05	2.0E-03	1.8E-05
Am-243	3.8E-12	6.1E-10	5.5E-12
Cm-243	9.0E-12	1.4E-09	1.3E-11
Cm-244	1.9E-12	3.1E-10	2.8E-12
Cm-245	4.7E-18	7.6E-16	6.8E-18
Cm-246	2.8E-21	4.5E-19	4.0E-21

A number of simplifying assumptions were made to bracket the isotopic activities in the Vycor glass waste stream. Shown in Table A-15 are the calculated lower-bound, best-estimate, and upper-bound estimates of contaminant profiles for the Vycor glass waste stream. The best-estimate activities were equated with the baseline activities in Table A-14. A number of factors contributed to uncertainties in calculating activities. Contributing factors included variations in fuel burnups, enrichments, reporting methods, and uncertainties in ORIGEN2 analysis. Because of limited information about this waste stream, a generic approach was taken. From Section 3.14 of this report, a generic uncertainty factor of 1.5 was employed to compute best-estimate and upper-bound activities associated with the Vycor glass waste streams. These uncertainty factors were applied to the baseline activities to estimate lower-bound, best-estimate, and upper-bound activities. The weights of the actinide residues are consistent with 10–20% losses in the leaching of and estimated 80 kg of uranium that was embedded in the Vycor glass scrap.

Table A-15. Net HDT isotopic distributions for Vycor glass sent to the SDA.

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
H- 3	2.2E+02	3.4E+02	5.0E+02
Be-10	9.4E-07	1.4E-06	2.1E-06
C-14	3.8E-05	5.7E-05	8.6E-05
Sr-90	2.8E+04	4.2E+04	6.4E+04
Nb-94	5.6E-06	8.4E-06	1.3E-05
Tc-99	4.4E+00	6.7E+00	1.0E+01
I-129	1.2E-02	1.8E-02	2.7E-02
Cs-137	3.1E+04	4.6E+04	6.9E+04
Eu-152	1.6E-01	2.5E-01	3.7E-01
Eu-154	2.7E+01	4.0E+01	6.0E+01
Pb-210	2.2E-10	3.2E-10	4.8E-10
Ra-226	5.4E-09	8.0E-09	1.2E-08
Ra-228	3.5E-14	5.2E-14	7.8E-14
Ac-227	3.0E-08	4.6E-08	6.8E-08
Th-228	2.4E-05	3.6E-05	5.3E-05
Th-229	6.6E-10	9.9E-10	1.5E-09
Th-230	6.3E-06	9.4E-06	1.4E-05
Th-232	2.1E-13	3.1E-13	4.7E-13
PA231	5.0E-07	7.5E-07	1.1E-06
U-232	3.3E-05	5.0E-05	7.5E-05
U-233	1.3E-06	1.9E-06	2.8E-06
U-234	1.8E-01	2.7E-01	4.0E-01
U-235	6.0E-03	9.0E-03	1.4E-02

Table A-15. (continued).

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
U-236	1.2E-03	1.8E-03	2.6E-03
U-238	8.5E-04	1.3E-03	1.9E-03
Np-237	8.8E-05	1.3E-04	2.0E-04
Pu-238	1.2E-02	1.7E-02	2.6E-02
Pu-239	4.7E-01	7.1E-01	1.1E+00
Pu-240	3.5E-03	5.3E-03	7.9E-03
Pu-241	2.1E-03	3.2E-03	4.7E-03
Pu-242	8.9E-11	1.3E-10	2.0E-10
Pu-244	1.0E-21	1.6E-21	2.3E-21
Am-241	1.2E-05	1.8E-05	2.7E-05
Am-243	3.7E-12	5.5E-12	8.2E-12
Cm-243	8.6E-12	1.3E-11	1.9E-11
Cm-244	1.9E-12	2.8E-12	4.2E-12
Cm-245	4.5E-18	6.8E-18	1.0E-17
Cm-246	2.7E-21	4.0E-21	6.0E-21

A-6. INTEC ONE-TIME NRF TEST CONTAMINANTS AND OTHER RELATED STREAMS

This section augments the reassessment analysis of the experimental contaminant byproducts documented in Section 3.7 of this report. These contaminant byproducts were the result of various experiments and tests conducted at INTEC during the HDT period. One byproduct waste stream was identified as emanating from a Navy-sponsored ETR experiment (LMITCO 1995). This waste stream consisted of significant amounts of fission contaminants along with actinide contaminant residues and some possible activation products. However, as shown in Section 3.7 it was determined that this particular waste stream had no more Cs-137 per unit weight than other similar irradiated actinide streams identified in Section A-4. The best-estimate amount of Cs-137 for this shipment was reassessed to be 7 Ci. This shipment was only a fraction of the total 5.4E2 Ci Cs-137 associated with irradiated actinide material sent to the SDA.

For the purpose of the reassessment analysis, it was assumed that Cs-137 was the dominant gamma-emitting isotope measured in most waste shipments containing irradiated actinide materials.[g] It was assumed that sufficient time had passed to allow other shorter half-lived gamma emitting isotopes to become insignificant. To reassess the nonuranium contaminant profiles in this stream, a set of calcine

g. There was one exception to this generality. In 1969, a one-time only disposal reported net gamma plus beta activities of 4E4 Ci. Most of this activity turned out to be short half-lived isotopes. The irradiated specimen was fresh and had little time to cool down.

based scaling factors was used. The updated contaminate distributions for this waste stream used an assumed isotopic profile matching zirconium-bearing calcine (Wenzel 2000b). The isotopic profile of interest was based on ORIGEN2 analysis. Shown in Table A-16 are specific activities and the associated scaling factors. All isotopes were scaled to Cs-137.

Table A-16. Zirconium-bearing calcine specific activities and scaling factors.

Nuclide	Specific Activity (Ci/g)	Scaling Factor Referenced to Cs-137
H-3	6.2E-06	3.3E-03
Be-10	6.3E-14	3.4E-11
C-14	2.6E-12	1.4E-09
Cl-36	—	—
Co-60	2.3E-06	1.2E-03
Ni-59	—	—
Ni-63	1.1E-06	5.6E-04
Sr-90	2.5E-03	1.3E+00
Nb-94	4.8E-13	2.6E-10
Tc-99	3.1E-07	1.7E-04
I-129	5.1E-10	2.7E-07
Cs-137	1.9E-03	1.0E+00
Eu-152	2.0E-07	1.1E-04
Eu-154	3.7E-05	2.0E-02
Pb-210	2.7E-18	1.5E-15
Ra-226	3.2E-17	1.7E-14
Ra-228	1.3E-18	7.0E-16
Ac-227	3.6E-13	1.9E-10
Th-228	1.4E-11	7.2E-09
Th-229	1.1E-16	5.9E-14
Th-230	1.8E-14	9.3E-12
Th-232	4.6E-18	2.4E-15
U-232	7.9E-14	4.2E-11
U-233	8.3E-16	4.4E-13
U-234	1.1E-08	5.8E-06
U-235	7.6E-11	4.0E-08
U-236	2.0E-10	1.1E-07
U-238	3.7E-12	2.0E-09

Table A-16. (continued).

Nuclide	Specific Activity (Ci/g)	Scaling Factor Referenced to Cs-137
Np-237	3.7E-10	1.9E-07
Pu-238	1.9E-05	1.0E-02
Pu-239	2.4E-07	1.3E-04
Pu-240	2.2E-07	1.2E-04
Pu-241	2.7E-05	1.4E-02
Pu-242	5.0E-10	2.6E-07
Pu-243	0.0E+00	0.0E+00
Pu-244	8.6E-23	4.6E-20
Am-241	1.1E-06	5.8E-04
Am-242	5.1E-13	2.7E-10
Am-243	2.9E-14	1.5E-11
Cm-242	4.2E-13	2.2E-10
Cm-243	1.4E-14	7.6E-12
Cm-244	1.9E-13	1.0E-10
Cm-245	2.4E-18	1.3E-15
Cm-246	3.7E-20	2.0E-17
Cm-247	9.0E-27	4.7E-24
Cm-248	1.8E-27	9.3E-25

A number of simplifying assumptions were made to bracket the isotopic activities in the above waste stream. Shown in Table A-17 are the calculated lower-bound, best-estimate, and upper-bound estimates of contaminant profiles for contaminants connected with irradiated uranium bearing actinides. Explicit shipments of U-234, U-235, and U-238 are reported in Section A-4. The lower-bound activities were equated with half the reported 5.4E2 best-estimate Cs-137 activity because of uncertainties in burnups of these disposed materials. A number of factors contributed to uncertainties in calculating best-estimate and upper-bound activities. Contributing factors included unknown burnups, enrichments, net activity reporting methods, and uncertainties in ORIGEN2 analysis. Because of these uncertainties, the upper-bound activity estimates was double the best-estimate activities.

Table A-17. Lower-bound, best-estimate, and upper-bound isotopic activity profiles for the 1969 Navy-sponsored reprocessing experiment waste stream.

Nuclide	Lower bound (Ci)	Best estimate (Ci)	Upper Bound (Ci)
H-3	8.9E-01	1.8E+00	3.6E+00
Be-10	9.1E-09	1.8E-08	3.7E-08
C-14	3.7E-07	7.3E-07	1.5E-06
Co-60	3.3E-01	6.5E-01	1.3E+00
Ni-63	1.5E-01	3.0E-01	6.0E-01
Sr-90	3.6E+02	7.1E+02	1.4E+03
Nb-94	6.9E-08	1.4E-07	2.8E-07
Tc-99	4.5E-02	9.0E-02	1.8E-01
I-129	7.4E-05	1.5E-04	2.9E-04
Cs-137	2.7E+02	5.4E+02	1.1E+03
Eu-152	2.9E-02	5.8E-02	1.2E-01
Eu-154	5.3E+00	1.1E+01	2.1E+01
Pb-210	3.9E-13	7.9E-13	1.6E-12
Ra-226	4.6E-12	9.2E-12	1.8E-11
Ra-228	1.9E-13	3.8E-13	7.6E-13
Ac-227	5.1E-08	1.0E-07	2.1E-07
Th-228	2.0E-06	3.9E-06	7.8E-06
Th-229	1.6E-11	3.2E-11	6.4E-11
Th-230	2.5E-09	5.0E-09	1.0E-08
Th-232	6.6E-13	1.3E-12	2.6E-12
U-232	1.1E-08	2.3E-08	4.5E-08
U-233	1.2E-10	2.4E-10	4.8E-10
U-234	—	—	—
U-235	—	—	—
U-236	2.9E-05	5.7E-05	1.1E-04
U-238	—	—	—
Np-237	5.3E-05	1.1E-04	2.1E-04
Pu-238	2.7E+00	5.5E+00	1.1E+01
Pu-239	3.5E-02	7.0E-02	1.4E-01
Pu-240	3.2E-02	6.4E-02	1.3E-01
Pu-241	3.9E+00	7.7E+00	1.5E+01

Table A-17. (continued).

Nuclide	Lower bound (Ci)	Best estimate (Ci)	Upper Bound (Ci)
Pu-242	7.1E-05	1.4E-04	2.9E-04
Pu-243	0.0E+00	0.0E+00	0.0E+00
Pu-244	1.2E-17	2.5E-17	4.9E-17
Am-241	1.6E-01	3.2E-01	6.3E-01
Am-242	7.3E-08	1.5E-07	2.9E-07
Am-243	4.2E-09	8.4E-09	1.7E-08
Cm-242	6.1E-08	1.2E-07	2.4E-07
Cm-243	2.1E-09	4.1E-09	8.3E-09
Cm-244	2.7E-08	5.4E-08	1.1E-07
Cm-245	3.5E-13	7.0E-13	1.4E-12
Cm-246	5.4E-15	1.1E-14	2.2E-14
Cm-247	1.3E-21	2.6E-21	5.2E-21
Cm-248	2.5E-22	5.1E-22	1.0E-21

A-7. CPP-603 BASIN SLUDGE DISPOSALS

This section augments the reassessment analysis of CPP-603 waste disposals sent to the SDA and documented in Section 3.9 of this report. This stream consisted of significant amounts of fission contaminants, actinide contaminants, and unknown amounts of activation products. These waste disposals were the consequence of an intensive cleanup campaign of the CPP-603 storage basins in 1976. The sludge was subsequently processed and shipped to the SDA during the period from 1977 through 1980 in 41 separate shipments.[h] To reassess the nuclide content of this waste stream the actual weight of the disposed dried sludge was reviewed. Assay data taken before the basin cleanup was then applied to this net disposal weight to reassess and calculate net amounts of U-235 and Cs-137. By using appropriate scaling factors referenced to Cs-137 and U-235 remaining contaminants were calculated for lower-bound, best-estimate, and upper-bound activities.

To better understand the background of the sludge cleanup campaign some basic background about the CPP-603 storage basin system is required. The basin complex is also referred to as the Fuel Receiving and Storage Facility. Before fuel reprocessing at INTEC, spent fuel is temporarily stored in this basin area. The storage basins were constructed in 1951 in an interconnected "E" shaped configuration that are designated as the south, middle, and north basins. The basins were filled with water with approximately 20 ft of shielding cover above the tops of the stored fuel.[i]

h. Two vaults shipped in 1977 and two others shipped in 1980 were under filled because of procedural errors. These other four shipments were reported in RWMIS and also mentioned in (Hoech and Rhodes 1979).

i. All fuel from CPP-603 were removed around 2000 and transferred to the CPP-666 complex. This facility is currently slated for D&D.

Since the basin operations from 1951 to 1976, accumulations of suspended dirt, dead algae, iron oxide, and other debris have built up a 5- to 10-cm sludge layer on the bottom of the basins.[j] Some of this debris contained radionuclides from fuel corrosion products that built up over approximately 26 years of operation (Smith 1974). Other particulate contaminants were released into the storage basins from sawing off ends of fuel assemblies before reprocessing. No analytical information was identified to evaluate actual amounts of saw fines or other cutting debris that was embedded in the basin sludge. In one instance, a fuel end plate was vacuumed up during sludge removal operations in the south end of the basin (Hoech and Rhodes 1979).[k]

Also, there were several events involving the rupturing of EBR-II fuel canisters (Low 1982) and (Smith 1974). The first event occurred in 1969, and resulted in a significant increase in basin water activities. Since EBR-II fuel is sodium bonded, the entry of water into a fuel storage canister results in the catastrophic failure of the stored fuel (DOE 1996). In these events, a chemical reaction between the sodium bond and water produces highly corrosive sodium hydroxide. Typically, the corroded bottom of the canister ruptures and fuel scrap is released to the bottom of the basin.

In addition, during cleanup operations it was discovered that radioactive ion-exchange resins had also leaked onto the basin floor. The water basin ion exchange system had a leaking retaining screen. A considerable amount of resin was released to the basin floor before the problem was corrected. Secondly, during the south basin cleanup routine, sampling showed in some zones around the cutting area that concentrations of U-235 exceeded the criticality safety limit of 0.145% weight of removed sludge (Hoech and Rhodes 1979). This would suggest that some cutting operations had inadvertently breached the fuel pins releasing fuel fragments to the basin floor. On the average, U-235 concentrations in the south basin were twice as high as in the north and middle basins.

Actual cleanup of the storage basins was done in several stages. Removal of accumulated basin sludge employed a vacuum cleaner system that transferred this material into a temporary holding tank. Later, this holding tank sludge was transferred into concrete vaults that were sent to the SDA for burial. Basin sludge was vacuumed up and stored in an intermediate 25,000-gal storage tank. Then the sludge was pumped into steel lined concrete storage vaults for final disposal. The vault containers were 55 in. in diameter by 55 in. high and with 18.5 in. of concrete shielding. The empty containers including shielding were cube shaped—with 7.5 ft (2.3 m) sides, and weighing 18.144 metric tons (Hoech and Rhodes 1979). Each concrete vault was reported to have a 12-gauge mild steel liner. The concrete shielding was necessary to reduce the radiation fields at the surface of the filled containers to less than 200 mR/h. The radiation from the unshielded sludge was of the order of 100 R/h.

After the sludge was pumped into each vault container the sludge was dewatered and solidified. Approximately 1.235 m^3 of sludge slurry was pumped into each vault. After each container was about two-thirds filled, additional chemicals were added to the remaining one-third of the container volume to facilitate solidification of the sludge. To accomplish this end, each vault was injected with 650 liters of urea-formaldehyde and mixed with 65 liters of 30% sulfuric acid. Air was then injected into this mixture to dewater the slurry by a moisture removal vent system. The chemical additives catalyzed the slurry into a dried solid form. It was estimated that 30% of the solidified weight in each vault was from the above-mentioned chemical additives. The average weight of the dried sludge per container was estimated from information presented in Hoech and Rhodes (1979). The density of the sludge slurry was estimated as 2.1

j. Originally, the basins were covered with steel gratings that corroded and released iron oxide to the basin bottoms. After the 1976 sludge cleanup, the gratings were replaced with fiberglass materials.

k. The south basin end was area used for fuel assembly cutting activity.

gm/cm^3 with a 30% volume reduction when dried. From the above volumetric fill data, 1.8–2.0 tons of dried basin sludge was contained in an average vault shipment. For the 41 reported shipments, the predicted net weight of disposed basin sludge is 74–80 metric tons.

A summary of RWMIS sludge disposals is presented in Table A-18. From this table, the net reported sludge disposals amounted to 150 metric tons of sludge along with chemical additives were sent to the SDA. The total weight of the sludge disposals was calculated by subtracting the nominal weight of the container (18.144 tons) from the reported gross shipping weights. From the above discussions about solidification treatments, it was estimated that 10–30% of the sludge was composed of dewatered chemical additives. This weight correction to Table A-18 would suggest that between 100 to 130 tons of dried sludge were actually shipped to the SDA.[1] However, the weights of the reported disposed sludge versus the above theoretical estimate of 74–80 tons differ significantly. The maximum uncertainty ratio for these weight differences was about 1.8. Reasons why the reported and predicted sludge weights differ include: some disposals were possibly not completely dried out, some shipments had higher than predicted weights because of the presence of gravel and other objects embedded in the sludge, and possible reporting errors in the weights of the disposals.

Table A-18. Summary of CPP-603 basin sludge cleanup disposals sent to the SDA.

Year	Number of Shipments	Total Activity (Ci)	Total Cs-137 Activity (Ci)	Total Shipping Weight (metric tons)	Total Reported Weight of Sludge (metric tons)
1977	9	3.9E+03	2.8E+02	2.1E+02	2.9E+01
1978	4	5.6E+03	2.0E+03	8.9E+01	1.7E+01
1979	2	1.0E+03	5.4E+02	4.3E+01	6.6E+00
1980	26	1.6E+04	9.4E+03	5.7E+02	9.8E+01
Total	41	2.6E+04	1.2E+04	9.1E+02	1.5E+02

The above reported and predicted estimates of actual sludge disposal upper and lower bounds for net contaminant disposals can be calculated using known assay data (Smith 1974). Assay data from the Smith report reported averaged contaminant profiles for both Cs-137 and U-235 on activity per unit weight basis (specific activities). By combining the reported net weights with the specific activities, net activities for Cs-137 and U-235 were calculated and compared with previously reported estimates. The bounding values for Cs-137 and U-235 activities are discussed below.

In 1973, 450 uranium samples were taken from the CPP-603 storage basins. The corresponding north, middle, and south basins each had 150 samples taken. Because of large number of samples, it was concluded that these data were statistically reliable. The percentage weight concentrations for the south, middle, and north basins were 0.070, 0.37, and 0.027 respectively (Smith 1974). The higher concentrations in the south basin are postulated to be the result of fuel subassembly removal hardware in that area and leaking fuel canisters. The average basin uranium sludge concentration was therefore 0.045%. Additional samples were taken in 1976 before the sludge removal. However, only thee samples were made, consequently this data was not considered as being statistically meaningful. The measured U-235 enrichment fractions varied from 46 to 26%. This suggested that leakage losses were from fuel other than highly enriched fuel from TRA and elsewhere. The enrichment assays were a closer match to

1. Because vault volumes constraint size is 1.95 m3 and the maximum solidified density is 2.8 metric tons per m^3 the physical limiting weight of all 41 shipments is approximately 224 metric tons. In practice the actual weight is lower.

early EBR-II fuels having U-235 enrichments from 48 to 52% before being burned (these fuels had 1–6 % burnups). However, these assays are partially consistent in the context that stored EBR-II fuel suffered several significant leakage events.

The lower bound, best estimate, and higher bounds for total U-235 activities shipped to the SDA are presented in Table A-19. The assumed uranium concentration was based on the average of 0.045% weight fraction for all sludge disposal. The lower and upper bounds were calculated by varying the assumed enrichments and total weights of the disposed sludge shipments. The best estimate was the average of the upper- and lower-bound activities. Another 50% uncertainty was factored into the best estimate to account for reporting uncertainties, and another 100% uncertainty was factored into the upper bound. These additional uncertainties account for limited isotopic assay information for 1976. That is, the 1973 data do not account for additional basin contaminant buildups up to 1976 when the basin sludge removal campaign began. Because of the relatively low U-235 enrichment of the measured basin samples, it was assumed that the nearest isotopic profile that was representative of the complete actinide profile would be similar to spent EBR-II fuel (see Section 3.6). The reassessed best estimate for net U-235 disposals amounted to about 16 kg; in contrast to the HDT best estimate, U-235 was 3 kg. It is unclear if localized hot spots may have skewed the measured uranium concentration basin distribution (of the 450 samples taken).

Table A-19. Lower bound, best estimate, and upper bounds for U-235 activities in sludge disposals sent to the SDA.

Bounding Size	Total wt of Sludge Shipments (metric tons)	Total Uranium (kg)	%U235 Enrichment	Total U-235 (kg)	Other Uncertainties	Total U-235 Activity (Ci)
Lower bound	77	3.4E+01	26	8.9E+00	1	1.9E-02
Best estimate	96	4.3E+01	36	1.6E+01	1.5	5.3E-02
Upper bound	115	5.1E+01	46	2.4E+01	2	1.0E-01

Calculations for the associated fission contaminants connected with Cs-137 are now discussed. In 1973, Cs-137 samples were also taken from the three storage basins (Smith 1974). A total of nineteen samples were taken with the highest readings occurring in the south basin region where fuel cutting activity was conducted. The Cs-137 sample data are presented in Table A-20. The reported distribution of Cs-137 does not suggest the presence of any localized radiation "hot spots." The average Cs-137 concentration was 2.9E-02 Ci/kg with a large standard deviation of 1.85E-02 Ci/kg. The best-estimate Cs-137 was set at the average value of 2.88E-02 Ci/kg, the lower- and upper-bound estimates were at one plus or minus standard deviation from the average. The lower- and upper-bound Cs-137 concentrations were 1.0E-02 Ci/kg and 4.7E-02 Ci/kg. To calculate the net Cs-137 activity disposals along with the attendant upper- and lower-bound values and approach similar to estimates for U-235 (see above Table A-19) were made. Table A-21 presents the lower-bound, best-estimate, and upper-bound estimates for Cs-137. Calculating the totals involved multiplying concentrations by total sludge weights with other additional uncertainties factored in. Another 50% uncertainty was factored into the best estimate to account for reporting uncertainties, and another 100% uncertainty was factored into the upper bound. These additional uncertainties account for reporting uncertainties such as missing information for basin contaminant buildup from 1973 through 1976.[m] The best-estimate activity of 4.2E3 Ci of Cs-137 was significantly smaller than the previously reported 1.5E4 Ci. The estimation methods for reporting the

m. Radionuclide assay data for 1996 was very limited.

RWMIS nuclide distributions were unknown. It is possible that measured gross gamma activities for the sludge disposals contained significant amounts of Co-60 that were possibly reported as Cs-137.

Table A-20. Measured Cs-137 activity data taken from the CPP-603 fuel storage basin.

Sample Location	Activity (Ci/kg)
North Basin	3.3E-02
North Basin	3.5E-02
North Basin	9.8E-03
North Basin	1.0E-02
North Basin	8.4E-03
Middle Basin	3.1E-02
Middle Basin	1.8E-02
Middle Basin	1.5E-02
Middle Basin	2.0E-02
Middle Basin	1.5E-02
South Basin	5.1E-02
South Basin	4.2E-02
South Basin	4.0E-02
South Basin	1.0E-02
South Basin	3.7E-02
South Basin	7.9E-02
South Basin	3.1E-02
South Basin	5.0E-02
South Basin	1.4E-02
Average	**2.9E-02**
Standard deviation	**1.9E-02**

Table A-21. Lower-bound, best-estimate, and upper-bound Cs-137 activities in sludge disposals sent to the SDA.

Bounding Size	Total wt of Sludge Shipments (metric tons)	Cs-137 Concentration (Ci/kg)	Other Uncertainties	Total Cs-137 Activity (Ci)
Lower bound	77	1.0E-02	1	8.0E+02
Best estimate	96	2.9E-02	1.5	4.2E+03
Upper bound	115	4.7E-02	2	1.1E+04

In 1993, Co-60 concentrations were measured from nineteen reported samples. The magnitudes of these samples varied significantly (activities varied by two orders of magnitude). The Co-60 basin activities are presented in Table A-22. Four of the data points (shown in bold) appeared to be possible localized "hot spots." The presence of these four points caused significant skewing of the data, as shown in the standard deviation calculated in the second column. The maximum Co-60 inventory was estimated with the hot spots included in Table A-22. The best-estimate and minimum activities were estimated using the second column in Table A-22. The lower bound was estimated as one standard deviation lower than the best-estimate value. Data in Tables A-22 and A-23 suggest that Co-60 was not a significant fraction of the best-estimate Cs-137 activity, but is a significant fraction for the upper-bound estimate.

Table A-22. Measured Co-60 sludge activities taken from the CPP-603 fuel storage basin.

Sample Location	Co-60 Activity (Ci)	Co-60 Activity Without Hot Spots (Ci)
North basin	8.08E-03	8.08E-03
North basin	**1.42E-01**	—
North basin	1.92E-03	1.92E-03
North basin	2.89E-03	2.89E-03
North basin	**2.73E-02**	—
Middle Basin	5.97E-03	5.97E-03
Middle Basin	9.46E-03	9.46E-03
Middle Basin	2.58E-03	2.58E-03
Middle Basin	5.16E-03	5.16E-03
Middle Basin	4.57E-03	4.57E-03
South Basin	7.41E-03	7.41E-03
South Basin	3.27E-03	3.27E-03
South Basin	2.09E-03	2.09E-03
South Basin	5.78E-04	5.78E-04
South Basin	1.93E-03	1.93E-03
South Basin	9.81E-03	9.81E-03
South Basin	6.46E-03	6.46E-03
South Basin	**9.65E-02**	—
South Basin	**1.12E-02**	—
Average	**1.8E-02**	**4.8E-03**
Standard deviation	**3.7E-02**	**2.8E-03**

Table A-23. Lower bound, best estimate, and upper bound for Co-60 activities in sludge disposals sent to the SDA.

Bounding Size	Total wt of Sludge Shipments (metric tons)	Co-60 Concentration (Ci/kg)	Other Uncertainties	Total Cs-60 Activity (Ci)
Lower bound	77	2.0E-03	1	1.5E+02
Best estimate	96	4.8E-03	1.5	6.9E+02
Upper bound	115	1.8E-02	2	4.2E+03

To estimate the total contaminant levels in the sludge disposals, the following assumptions were made. All fission products were scaled to the calcine nuclide distribution. All actinides and decay daughter products were scaled to the EBR-II Mark I nuclide distributions given in Section A-5. These EBR-II scaling factors were used because of reported canister failures of EBR-II fuel and because assayed sludge uranium enrichments approximated those of EBR-II fuels. Measured Co-60 assays showed the presence of activation products in the sludge. Since other zirconium and aluminum bearing fuels only contain trace amounts of Co-60 activity, it was assumed that the stainless steel bearing EBR-II fuel were a likely contributor to these contaminants (again because a disproportionate amount of fuel leakage). Scaling factors from Appendix C were used to calculate other activation contaminants of interest such as Ni-63.

Using the limiting activities from Tables A-21, A-23, and A-24, the corresponding lower-bound, best-estimate, and upper-bound isotopic distributions were calculated in Table A-24. The fission product distribution in Table A-24 was based on the calcine scaling factors, the actinides from Section A-5, and the activation products from Appendix C. The fission products were scaled to the bounding activities for Cs-137. The actinides were scaled to the limiting activities for U-235. The activation products were scaled to the bounding activities for Co-60.

Table A-24. Bounding HDT contaminant distributions for CPP-603 sludge disposals sent to the SDA.

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
H-3	2.3E+00	1.2E+01	3.1E+01
Be-10	3.0E-08	1.6E-07	4.1E-07
C-14	2.5E-03	1.1E-02	6.9E-02
Cl-36	1.4E-06	6.1E-06	3.7E-05
Co-60	1.6E+02	7.0E+02	4.2E+03
Ni-59	1.3E-02	6.2E-02	3.3E-01
Ni-63	9.9E-01	4.6E+00	2.5E+01
Sr-90	8.1E+02	4.2E+03	1.1E+04
Nb-94	5.7E-04	2.6E-03	1.6E-02
Tc-99	1.6E-01	8.2E-01	2.2E+00
I-129	2.5E-04	1.3E-03	3.5E-03
Cs-137	8.0E+02	4.2E+03	1.1E+04
Eu-152	5.6E-02	2.9E-01	7.7E-01
Eu-154	1.0E+01	5.4E+01	1.4E+02
Pb-210	6.9E-10	1.9E-09	3.7E-09
Ra-226	1.7E-08	4.7E-08	9.1E-08
Ra-228	1.1E-13	3.1E-13	5.9E-13
Ac-227	9.8E-08	2.7E-07	5.2E-07
Th-228	7.7E-05	2.1E-04	4.0E-04
Th-229	2.1E-09	5.8E-09	1.1E-08

Table A-24. (continued).

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
Th-230	2.0E-05	5.5E-05	1.1E-04
Th-232	6.7E-13	1.8E-12	3.6E-12
Pa-231	1.6E-06	4.4E-06	8.6E-06
U-232	1.1E-04	2.9E-04	5.7E-04
U-233	4.1E-06	1.1E-05	2.1E-05
U-234	5.7E-01	1.6E+00	3.0E+00
U-235	1.9E-02	5.3E-02	1.0E-01
U-236	3.8E-03	1.0E-02	2.0E-02
U-238	2.7E-03	7.5E-03	1.4E-02
Np-237	2.9E-04	7.8E-04	1.5E-03
Pu-238	3.8E-02	1.0E-01	2.0E-01
Pu-239	1.5E+00	4.2E+00	8.1E+00
Pu-240	1.1E-02	3.1E-02	6.0E-02
Pu-241	6.8E-03	1.9E-02	3.6E-02
Pu-242	2.9E-10	7.8E-10	1.5E-09
Pu-244	3.3E-21	9.1E-21	1.8E-20
Am-241	3.8E-05	1.0E-04	2.0E-04
Am-243	1.2E-11	3.2E-11	6.2E-11
Cm-243	2.8E-11	7.6E-11	1.5E-10
Cm-244	6.0E-12	1.6E-11	3.2E-11
Cm-245	1.5E-17	4.0E-17	7.7E-17
Cm-246	8.7E-21	2.4E-20	4.6E-20
Cm-247	0.0E+00	0.0E+00	0.0E+00
Cm-248	0.0E+00	0.0E+00	0.0E+00

A-8. TANK FARM CONTAMINATED SOIL DISPOSALS

This section augments the reassessment analysis of Tank-Farm-related soil disposals sent to the SDA that were documented in Section 3.10 of this report. This stream consisted of significant amounts of fission contaminants along with actinide contaminant residues and trace amounts of activation products. These waste disposals were the consequence of inadvertent subsurface releases of liquid waste in the Tank Farm storage area. None of these releases were caused by actual breaches in HLW storage tanks, but rather leaks from adjoining raffinate transfer lines. The Tank Farm complex consisted of eleven 300,000-gal tanks. Those tanks were used for storing radioactive liquid waste generated from reprocessing spent nuclear fuel (Palmer 1998).

Disposals of contaminated Tank Farm soil are summarized in Table A-25. Table A-25 presents net activities along with corresponding volumes and weights of disposed soil. Approximately 5.7E3 Ci was

sent to the SDA from 1974 to 1983, consisting of 3.6E3 metric tons of soil. From this table it is clear that the bulk of the disposed of activity was clustered in the years from 1974 through 1975. However, the reported bulk volumes and weights do not directly correlate with activity disposals. This was because some smaller secondary spills had disproportionately larger volumes of soil removed due to more extensive excavation cleanup work. In some circumstances, "hotter spills" resulted in only partial excavations of released tank farm waste because of the high attendant radiation fields and safety constraints on excavation activity. Quantities presented in Table A-25 were compiled from the RWMIS data. The reported INTEC locations of soil disposals sent to the SDA were not directly referenced to a particular ground release ground site, but rather to one of the buildings surrounding the Tank Farm complex. Consequently, the soil disposal data could not be directly connected to a specific release event for years where there were multiple releases or where cleanup excavations were staggered over multiple years.

Thus other supporting documentation was needed to clearly correlate a reported ground leakage event with actual disposal data (LMITCO 1995; WINCO 1993). Generally, when the removed contaminated soil had high associated gamma fields precautions were made to limit radiation exposure to the workers involved in a spill cleanup. One main precaution involved mixing contaminated soil with uncontaminated soil before shipment to the SDA. This soil mixing reduced the attendant radiation fields of the shipments. Consequently, reported RWMIS volumes and weights of the reported soil disposals were in some cases substantially larger than reported amounts of soil actually excavated from a specific spill sites. RWMIS data showed that soil shipments were made using Dempster-Dumpsters, cardboard boxes, and some cases wooden boxes. Generally, these containers were lined with plastic and sealed with duct tape (LMITCO 1995).

Table A-25. Summary of Tank Farm leak cleanup soil disposals to the SDA during the HDT.

Year	Gross Volume (m^3)	Gross Weight (metric tons)	Gross Activity (Ci)
1974	3.3E+02	6.4E+02	3.9E+03
1975	7.7E+01	4.6E+01	1.6E+03
1976	1.7E+02	9.6E+01	5.0E+01
1977	7.1E+01	9.7E+01	1.1E+02
1978	4.3E+02	2.0E+02	2.6E+01
1983	2.7E+03	2.5E+03	2.6E+01
Total	3.8E+03	3.6E+03	5.7E+03

From known shipping data coupled to reports dealing with accidental Tank Farm ground leakage events, it was concluded that about 97% of the activity removed from cleanup operations was associated with two events. The location of the first event was designated as site CPP-27 and was discovered in 1974; the second event designated as site CPP-28. Both of these sites involved leaks from subsurface pipes that transfer HLW liquids to the Tank Farm complex.

Site CPP-27 involved a 3-in. transfer subsurface leak that involved a release of 1,000–3,000 Ci of released gross activity. The amount of released HLW was estimated to less than 100 gal. It was suspected that this line had been leaking since 1961. A total of 210 m^3 of contaminated dirt were reportedly removed from this cleanup site and sent to the SDA. It was estimated that after the cleanup in the mid-1970s, that only small residual amounts of activity remained in the soil (25 mCi). However, in 1983 additional contaminated soil was detected; judged to be a second release from the same transfer line. Cleanup of this second spill resulted in the removal of approximately 1.1E4 m^3 of soil with 1.5E3 m^3

shipped to the SDA. The remainder of the contaminated dirt was sent to a burial site at INTEC. From RWMIS records and Form 110 shipping manifests, it was determined that 1983 cleanup involved substantially less activity sent to the SDA compared to the 1974 cleanup disposals to the SDA. However, because of a transition in reporting methods, content codes for disposals were not clear. Consequently, shipments of large volume noncompactable waste were assumed to contaminated soil from the 1983 cleanup campaign. The estimated gross activity sent to the SDA in 1983 was about 26 Ci (see Table A-25).

Contaminated soil associated with site CPP-28 was also discovered in 1974. The leak was associated with a 3-in. transfer line used to carry first-cycle raffinates. Leakage was caused by a 1/8-in. hole that was suspected to have been present during its initial installation in 1955. After this leak was discovered, a partial cleanup and excavation was conducted. A reported 43 m^3 of contaminated dirt were sent to the SDA that contained about 3.0E3 Ci of gross activity. It was reported that approximately 3.6E3 gal of HLW liquids had been released into the soil around the CPP-28 site. This volumetric release corresponds to about 3.2E4 Ci of gross gamma activity; this estimate was made by assuming a release of 9 Ci per gal of Cs-137 activity (WINCO 1993).

To reassess actual activity inventories associated with the Tank Farm leaks, the reported isotopic distribution in the contaminated soil were first reviewed. These distributions were reported on Form 110 shipping manifests, repeated in RWMIS, and in the HDT inventory report (LMITCO 1995). The review showed that important COCs such as Tc-99, I-129, and most actinides were not generally reported. Secondly, the reported fission product isotopic distributions were not representative. That is, the reported nuclide distributions were not representative of fission products found in raffinates (Palmer 1998), fuels (Wenzel 2000a), or calcine material (Wenzel 2000b). An example of how isotopic distributions were skewed is shown in Table A-26 for activity ratios of Cs-137 to Sr-90. The bottom entry is the reported ratio that is off by more than a factor 2 from other known fission distributions. Additionally, other similar inconsistencies for other isotopic ratios were also identified which are not explicitly discussed in this report.

Table A-26. Comparisons of Cs-137/Sr-90 ratios for typical INTEC high level waste versus reported RWMIS ratios.

Ratio Type	Cs-137/Sr-90
Generic PWR fuel ratio	1.0E+00
Generic TRA fuel ratio	1.0E+00
Average Calcine ratio	9.9E-01
Average Tank Farm ratio	1.2E+00
Reported ratio in shipping manifest and RWMIS	4.1E-01

Comparison done for reported SDA disposals in 1974–1975 time frames.

Because of the above-noted limitations in reported isotopic profiles for the contaminated soil waste stream, the contaminant profile was recalculated. To accomplish this end, an appropriate representative isotopic distribution was chosen for the contaminated soil disposals. It was judged that the isotopic distribution for the averaged calcine waste stream present in Section 3.8 of this report was representative of the liquid HLW since the calcine is physically reprocessed raffinates that are transformed into solid calcine material. These scaling factors were referenced to Cs-137. The activity for the CPP-28 release site had specific activity of 9 Ci per gal of spilled HLW liquid (WINCO 1993). This specific activity was previously applied to the CPP-28 spill in other reported analysis. The identical specific activity was applied to the CPP-27 site release.

To estimate contaminant inventories, net volumes of excavated soil were correlated with equivalent volumes of spilled raffinates embedded in those soil shipments. An equivalent amount of Cs-137 was calculated for contaminated soil sent to the SDA. This equivalent amount of Cs-137 was then applied to the calcine scaling factors in Appendix E to calculate remaining contaminant activities. Shown in Table A-27 are estimated bounding values for actual volumetric disposals and equivalent Cs-137 from the principal spill sites. For each spill site, the lower-bound, best-estimate, and upper-bound estimates of total gal and corresponding net Cs-137 activities are shown. A rationale is now discussed for how these estimated source estimates were made for each spill site.

Table A-27. Estimated net embedded raffinates and net cesium activities sent to the SDA for contaminated Tank Farm soil.

	CPP-28 (gal)	CPP-27 (gal)	Total (gal)	Equivalent Cs-137 Total Activity (Ci)
Lower bound	7.8E+01	5.0E+01	1.3E+02	1.1E+03
Best estimate	2.2E+02	7.5E+01	3.0E+02	2.7E+03
Upper bound	9.0E+02	1.0E+02	1.0E+03	9.0E+03

For the CPP-27 cleanup it was reported that effectively all of the surrounding soil was excavated and sent to the SDA. The reported amount of the HLW liquid spill was not "less than 100 gal" (WINCO 1993). Using professional judgment, it was concluded that a lower bound of 50 gal, an upper bound of 100 gal, and a best estimate of 75 gal of soil embedded HLW were sent to the SDA from this cleanup site.

For site CPP-28 it was reported that 3,600 gal was released near the transfer line break location. It was reported that depths below approximately 6.5 ft were not excavated for soil removal because of high

radiation fields (it was thought that peripheral areas with low radiation readings were excavated to lower depths). During the cleanup about 20 ft of this transfer line (including the leaking section) were removed for inspection. After the partial cleanup, clean soil was backfilled into the excavated trench. From the description of the excavation it was concluded there was an uncertainty of least a half a foot relative to the 6.5 ft reference depth for the cleanup excavation. From borehole radiation measurements, the subsurface contaminants were concentrated in a very nonuniform distribution around the leak location. From borehole radiation readings in the immediate area of the release it was concluded that most of the activity remained in nonexcavated zones of the cleanup activity. Best-estimate analysis shows that about 6% of the total activity from this spill was sent to the SDA. Shown in Table A-28 are two borehole radiation measurements taken along axial locations near and above the subsurface leak site (WINCO 1993). These radiation readings are sharply peaked in the subsurface zone from 6 to 9 ft.

Table A-28. Radiation readings for the subsurface spill site CPP-28.

Depth (ft)	Borehole 4 (mR/hr)	Borehole 8 (mR/hr)
0	3.0E+01	3.0E+01
1	3.5E+01	4.0E+01
2	6.0E+01	5.0E+01
3	7.0E+01	7.0E+01
4	1.0E+02	8.0E+01
5	1.5E+02	1.0E+02
6	2.0E+02	3.5E+02
6.5	1.5E+03	2.0E+03
7	5.5E+03	7.0E+03
7.5	3.5E+04	1.2E+04
8	2.0E+04	9.0E+04
8.5	3.0E+03	6.5E+04
9	8.0E+02	1.0E+04
10	1.0E+02	1.0E+03
11	5.0E+01	1.2E+01
12	5.0E+01	4.0E+00
13	6.0E+01	2.0E+00

Using the datum line of 6.5 ft, the integrated radiation load above and below this reference depth with a 6-in.depth uncertainty showed that the lower-bound, best-estimate, and upper-bound estimates for soil embedded contaminant activities were equal to 7.8E+01, 2.2E+02, and 9.0E+02 gal of equivalent source activity , respectively. These estimates are based on a net spill of 3,600 gal and the integrated areas between 6 and 7 ft were proportioned to that reported volumetric release. The lower-bound excavated soil was at 6 ft, the best-estimate depth was at 6.5 ft, and the upper-bound depth was at 7 ft. The ratio of the excavated upper surface radiation load to the total radiation load was averaged for the two borehole readings in Table A-29. The volumetric source embedded in the shipped contaminated soil was calculated

from the product of 3,600 gal and the excavated radiation fraction from Table A-29. The corresponding CPP-28 volumetric source bounding estimates are shown in Table A-27.

With a specific activity of 9 Ci/gal for the volumetric sources in Table A-27, the corresponding equivalent Cs-137 contaminant activities were estimated. Applying these activities from Table A-27 to averaged calcine scaling factors, the corresponding contaminant distributions are presented in Table A-29 for the lower, best estimate and upper bounds, respectively. Other smaller secondary spills with excavated dirt sent to the SDA in the 1974 through 1983 period were also proportioned to this same specific activity level. These smaller shipments amounted to about a 3% correction to the net activities for the two major spills from sites CPP-27 and CPP-28.

Table A-29. Bounding and best-estimate HDT contaminant distributions for Tank Farm soil sent to the SDA.

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
H-3	3.4E+00	7.9E+00	2.7E+01
Be-10	4.5E-08	1.1E-07	3.5E-07
C-14	1.9E-06	4.4E-06	1.5E-05
Cl-36	—	—	—
Co-60	1.5E+00	3.5E+00	1.2E+01
Ni- 59	4.0E-03	9.3E-03	3.1E-02
Ni- 63	2.7E-01	6.3E-01	2.1E+00
Sr-90	1.2E+03	2.8E+03	9.4E+03
Nb-94	3.7E-07	8.6E-07	2.9E-06
Tc-99	2.3E-01	5.5E-01	1.8E+00
I-129	3.8E-04	8.9E-04	3.0E-03
Cs-137	1.2E+03	2.8E+03	9.3E+03
Eu-152	8.4E-02	2.0E-01	6.6E-01
Eu-154	1.5E+01	3.6E+01	1.2E+02
Pb-210	3.1E-09	7.2E-09	2.4E-08
Ra-226	4.4E-07	1.0E-06	3.4E-06
Ra-228	6.6E-11	1.5E-10	5.2E-10
Ac-227	1.1E-07	2.7E-07	8.9E-07
Th-228	2.0E-02	4.8E-02	1.6E-01
Th-229	9.4E-10	2.2E-09	7.4E-09
Th-230	1.6E-06	3.8E-06	1.3E-05
Th-232	3.1E-12	7.2E-12	2.4E-11
Pa-231	3.0E-07	7.0E-07	2.3E-06

Table A-29 (continued).

Nuclide	Lower Bound (Ci)	Best Estimate (Ci)	Upper Bound (Ci)
U-232	4.1E-05	9.6E-05	3.2E-04
U-233	5.9E-08	1.4E-07	4.6E-07
U-234	4.2E-03	9.7E-03	3.3E-02
U-235	2.7E-05	6.3E-05	2.1E-04
U-236	6.8E-05	1.6E-04	5.3E-04
U-238	1.4E-06	3.3E-06	1.1E-05
Np-237	4.1E-04	9.5E-04	3.2E-03
Pu-238	4.9E+00	1.2E+01	3.9E+01
Pu-239	8.1E-02	1.9E-01	6.4E-01
Pu-240	6.3E-02	1.5E-01	5.0E-01
Pu-241	7.2E+00	1.7E+01	5.7E+01
Pu-242	1.3E-04	3.1E-04	1.0E-03
Pu-244	2.0E-12	4.7E-12	1.6E-11
Am-241	3.5E-01	8.2E-01	2.8E+00
Am-243	3.3E-03	7.8E-03	2.6E-02
Cm-243	1.0E-04	2.5E-04	8.3E-04
Cm-244	6.4E-03	1.5E-02	5.1E-02
Cm-245	5.8E-07	1.4E-06	4.6E-06
Cm-246	4.6E-08	1.1E-07	3.6E-07
Cm-247	6.0E-14	1.4E-13	4.7E-13
Cm-248	7.3E-14	1.7E-13	5.7E-13

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